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Progress in Development of Fluoride-Salt-Cooled High-Temperature Reactors (FHRs)

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INTRODUCTION

Fluoride-salt-cooled High-temperature Reactors (FHRs) are a new class of reactors that use graphite-matrix coated-particle fuel with a clean liquid salt coolant. The largest university FHR program is an Integrated Research Project (IRP) sponsored by the U.S. Department of Energy and others that is led by the Massachusetts Institute of Technology (MIT) and includes the University of California at Berkeley (UCB), the University of Wisconsin (UW), and the University of New Mexico (UNM). The year 2016 was the fifth year of this program. We report herein some of the major results of work done in 2016.

The base-line FHR concept studied in this IRP [1-3] uses (1) graphite-matrix coated-particle fuel with a pebble bed geometry and (2) a clean liquid salt coolant—flibe (a mixture of ^{7}LiF and BeF_2). The combination of a very high-temperature fuel with failure temperatures above 1650°C, a coolant boiling point above 1400°C, and a coolant that traps most fission products implies the capability to assure no major fuel failure under severe accident conditions. There are a family of FHR concepts using different fuel geometries and different fluoride salts.

The power conversion system is a Nuclear Air-Brayton Combined Cycle (NACC) with peak power provided by natural gas or a Firebrick Resistance Heated Energy Storage (FIRES) system. This enables a base-load reactor to provide variable electricity to the grid that substantially increases revenue relative to base-load nuclear reactors.

Most of the results reported herein are applicable to most FHR concepts as well as liquid-fueled molten salt reactors (MSRs).

IRRADIATION EXPERIMENTS

The most recent FHR in-reactor experiment in MIT research reactor (MITR) was successfully installed and completed its irradiation testing in December 2016. Ninety five (95) test specimens were irradiated in 700°C flibe salt for ~1000 hours with an average neutron flux 2.4E14 n/cm^2-s (full energy range). This new flibe salt irradiation facility, FS-3, included internal electric resistance heating to assure temperature control above 200°C to reduce the generation of radiolysis products when the reactor was not operating as well as temperature control during reactor operations. The FS-3 test consists of a single nickel capsule containing three graphite crucibles. The system measure off-gas compositions from the capsule and tritium releases via diffusion through the capsule walls during the irradiation. The design and analysis of this facility (Fig. 1), described in previous reports, encompassed experts across a variety of fields including neutronics, materials, and thermal-hydraulics as part of the MITR's required in-core experimental safety evaluation report.

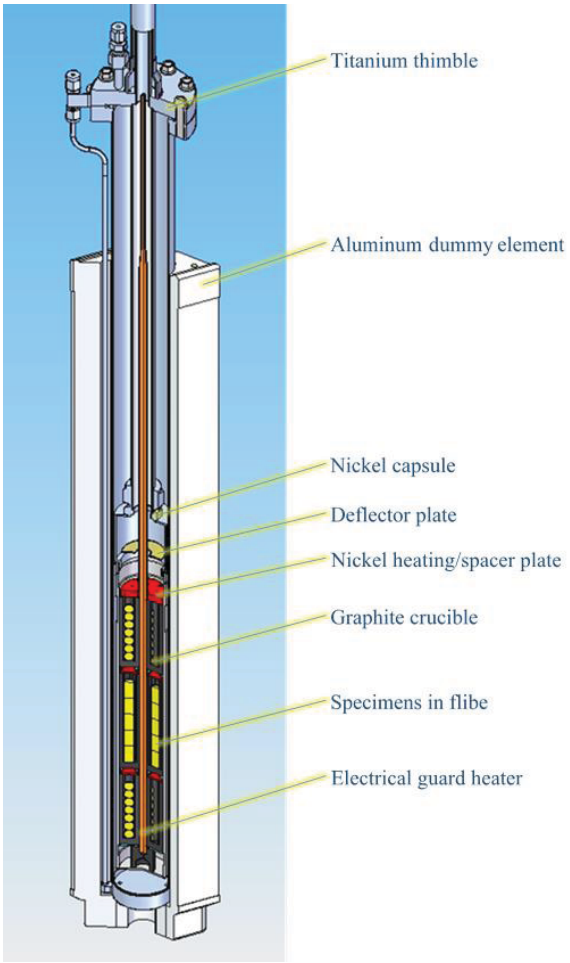

Fig. 1. FS-3 Experiment Assembly MITR Core Section

The specimens for this test were provided by UW, ORNL, INL, and the Chinese Academy of Sciences/Shanghai Institute of Applied Physics (SINAP), and included multiple types of nuclear-grade graphite and carbon/carbon composites.

Specimens were prepared and analyzed at both MIT and UW, including baking under a hydrogen environment to remove moisture and oxygen impurities. The irradiation crucibles were designed and assembled by the MIT Nuclear Reactor Laboratory (NRL), and then the graphite crucibles and specimens were shipped to UW for salt filling. Once successfully loaded and filled with precise volumes of salt, the crucibles were returned to the NRL for final drying and inerting under helium before being assembled into the instrumented irradiation capsule and inserted into the reactor. After completion of the tests, the assembly was removed from the reactor and placed in a hot cell to cool down before being disassembled.

Parallel corrosion tests using the same materials and salt under the same conditions are being done at UW to determine the effect of radiation on corrosion rates. The U.S. now has the demonstrated capability to conduct large-scale sample irradiations in 700°C flibe salt at the MITR, a capability that has not existed since the 1960s and with the addition of modern instrumentation and analytical tools that greatly increase the information obtained from irradiating samples.

CORROSION EXPERIMENTAL RESULTS

Large-scale corrosion tests for materials of construction were conducted in flibe salt at 700°C. This includes a major effort to manufacture, purify and characterize (redox and impurities) flibe at UW that is supplying salt for experimenters. Developments this year included better methods to measure salt impurities and measure redox—the parameter that determines the potential for corrosion of different materials in high-temperature salts. The redox measurement technology enables control of redox and thus the potential for corrosion and appears to be viable for both laboratory and use in an FHR.

There are two FHR materials environments: (1) in-core where materials can be corroded in salt with the combined effects of radiolysis and transmutation and (2) outside the reactor core with the greatly reduced radiation field such as in heat exchangers (in operating FHRs gamma radiation will be emitted by short-lived decay of ^{18}F and ^{20}F produced by neutron reactions on ^{19}F). Testing of materials in salt for the in-core environment is a joint effort by UW and MIT with in-core tests done in the MIT reactor (see above) and UW where testing outside of the irradiation field is done. Most of the testing of in-core materials is associated with graphite—the neutron moderator and a major component of the fuel. The compatibility of graphite with flibe was demonstrated in the Molten Salt Reactor Experiment (an 8 MWt test reactor) in the 1960s. The experimental effort to date is to understand the uptake of tritium and the use of new in core materials such as SiC-SiC composites—materials not available in the 1970s. The testing has shown that some types of SiC-SiC composites have good behavior in 700°C salt under irradiation but other types degrade and are unsuitable for reactor use.

Testing of metals of construction show Hastelloy-N has good behavior, as expected. Limited testing indicates that 316SS may be a suitable material of construction for the FHR with appropriate control of salt impurities and redox. There are large incentives to use 316SS because of its lower cost and because it is a nuclear code-qualified material of construction. A corrosion flow loop is starting up at UW that will enable investigation of 316SS corrosion under more realistic conditions of flowing salt and temperature differences around the circulating salt loop.

A method for electrochemical purification of flibe at UW was discovered in the development of the redox measurement systems [4]. Experiments are underway to see if this could be an alternative method to purify salts in the laboratory and ultimately in an FHR.

CONTROL OF TRITIUM

Liquid salts, particularly flibe, generate significant quantities of tritium under neutron irradiation that must be controlled and captured to avoid its release to the environment via heat exchangers. The combination of experiments and modelling [5, 6] in 2016 suggest a practical way to remove tritium to very low levels in liquid salt at 700°C salt using carbon beds. A carbon bed is a leading candidate for tritium removal from salt at 700°C because (1) it can sorb significant amounts of tritium as demonstrated in the molten salt reactor experiment where 15% of the tritium remained with the nuclear-grade carbon moderator and (2) carbon is chemically compatible with fluoride salts. Understanding tritium uptake on carbon is also required because FHRs use a graphite-matrix coated-particle fuel. If there is a large uptake of tritium on the fuel, that tritium could be released in an over-temperature transient. In the case of a pebble bed reactor where fuel and blanket pebbles are circulated through the reactor core, there may be the option of removing tritium from the salt using these pebbles.

Experimental measurements of hydrogen sorption and carbon properties were made on several classes of carbon: (1) nuclear grade graphites used in the fuel and (2) non-nuclear graphites. Carbon comes in many forms. The processing of graphite to withstand neutron irradiation with limited dimensional changes creates a carbon form that limits tritium uptake. If a carbon bed is used for tritium removal outside the reactor core, there is no requirement for the carbon bed to withstand high radiation fields. The experimental work this year indicates that tritium sorption on such non-nuclear-grade carbons relative to nuclear-grade graphites is several orders of magnitude better for similar-sized carbon-bed tritium removal systems and therefor a

practical efficient tritium removal system appears possible. The physical characterization of these different forms of carbon enables understanding of what characteristics favor tritium uptake and under what conditions.

The experimental work included measuring hydrogen uptake on carbon at very low hydrogen partial pressures (near 1 Pa) as would be required in an FHR, characterization of nuclear and non-nuclear grade carbons and measurement of tritium uptake by various graphites irradiated in 700°C salt. In addition, work is underway at the UNM on an alternative tritium removal method using gas sparging and ultrasonics to improve the rate of tritium transport from the liquid phase to the gas phase. Much of this work is also directly applicable to development of liquid-fueled molten salt reactors for tritium and noble metal removal from their fuel salt.

THERMAL HYDRAULICS AND NEUTRONICS

A major experimental program and modeling effort on FHR thermal hydraulics, neutronics, and structural mechanics, especially in relation to safety modeling of passive decay heat removal systems during transients and accidents, produced many results this year. Experimental data is used to validate models where possible, which will be part of a benchmark campaign involving all IRP members as well as outside participants. The coupling of models in thermal hydraulics, neutronics, and structural mechanics for a holistic view of FHR phenomena and response is the long-term goal.

UCB continues to operate its Compact Integral Effects Test (CEIT) facility [7], a 10-m high test facility that simulates the thermal hydraulics of an FHR using Dowtherm-A, a fluid with properties at less than 100°C that behaves similar to liquid salts at 700°C (Fig. 2). The CIET Facility underwent final adjustments in its current configuration which has produced significant data for the validation and verification of thermal hydraulic system models using codes including RELAP5-3D and FANCY.

Additional modifications and capability expansions are planned such that the CIET Facility can be controlled to behave as a small test-scale FHR. Additional separate-effect test work with simulant fluids was also performed to study heat transfer phenomena continues in order to understand higher-fidelity in-core behavior. Core design, neutronics models, and neutronics code-to-code and benchmarking activities are important parallel research branches to support overall understanding of FHR behavior, design, and safety.

A benchmark suite was created to perform neutronics code-to-code comparisons [8]. The benchmark includes tests that allows one to focus on specific phenomena (for example salt cross sections and fuel double heterogeneity) as well as more complex tests based on full core models. Multiple Monte Carlo codes (Serpent, MCNP, and KENO-VI) were compared using this benchmark set. The most complex cases show a discrepancy of up 400 pcm on the multiplication factor. Such discrepancy appears to be acceptable when compared with all sources of uncertainty.

UCB is also developing new capabilities to couple neutronics and thermal-hydraulics for modeling FHR transient scenarios. High-fidelity models have been developed coupling Serpent, for neutronics, with OpenFOAM, for thermal-hydraulics (porous media) and pebble motion (discrete element modeling) [9]. Phenomena such as pebble loading, control rods insertion, reactivity insertion, and more can be studied using this tool and detailed information can be obtained at the individual pebble and even at individual kernel level.

Such models are computationally expensive and are meant to be used mostly to guide the development and the validation of reduced order models. For this reason, a tool for rapid scoping calculations was developed leveraging the COMSOL Multiphysics software. COMSOL uses the finite element method for spatial discretization to solve systems of PDEs or ODEs, and allows the user to combine pre-defined multiphysics modules with a system of user-defined PDEs.

Fig.2. UCB CIET Facility

Unlike water and sodium, the physical properties of liquid salts are not as well known. Experiments and studies on the impact of uncertainties of liquid salt properties on the safe operational envelopes for pebble bed and prismatic core FHRs were completed at MIT for the baseline salt (flibe) and sodium zirconium fluoride salts.

Initial studies [10] examined the option of a subcritical FHR test facility coupled to the MIT reactor. The MITR was designed, licensed, and built with the capability to drive subcritical experiments. A subcritical test facility could operate a small fraction of an FHR core at about 30% the power density of a full-scale reactor with prototypical fuel and 700°C salt. This could provide a realistic test bed to rapidly collect data for a full scale reactor.

Work at MIT as begun on design of facilities to measure radiative heat transfer of liquid salts in the 600 to1100°C range. Because liquid salts are participating media and radiative heat transfer increases as the fourth power of the absolute temperature, this heat transfer mechanism becomes important under accident conditions.

EVALUATION MODEL BENCHMARKING AND VALIDATION

FHR Evaluation Model Benchmarking and Validation Workshops involve conducting joint workshops with IRP member universities, advisory committee members, and relevant experts to review current progress, help direct further efforts, and assure that the most critical issues of FHR technology development are addressed by the IRP. The second IRP Benchmarking Workshop conducted April 14-15, 2016 saw the formation of three separate working groups: materials activation, tritium, and transport (MATT); thermal hydraulics; and neutronics. Additional members from the international molten salt reactor community are engaged and work is progressing steadily throughout these areas.

The next workshop will be integrated with the Georgia Tech-led FHR IRP in March 2017 on *Tools for Modeling and Simulation of Fluoride Cooled High-Temperature Reactors (FHR) – Gaps and Development Needs*.

CONCLUSIONS

Major progress has been made in development of the technologies required for an FHR test reactor and its ultimate commercialization. This includes addressing the major technical issues and addressing the economics with a power cycle to meet the 2030 needs of the electrical grid [11].

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